



Ontario Hydro



RESEARCH DIVISION



**ontario hydro
research division**

**MATERIALS INTEGRITY SECTION
A REVIEW OF 1986 AND PROPOSALS FOR 1987**

Report No 87-36-K

R.G. Fleck
Supervising Engineer
Materials Integrity Section
Metallurgical Research Department

ABSTRACT

This report describes the work programs performed in 1986 by the Materials Integrity Section of the Metallurgical Research Department. The direction of the 1987 work programs is outlined. The main thrust of these programs will be in the area of CANDU core components with emphasis on the influence of hydrides on pressure tube cracking and the role of metallurgical structure on pressure tube deformation. The purpose of these studies is to develop a more reliable predictive capability as opposed to the extrapolation of in-service data, for planning reactor rehabilitation programs. In support of thermal plant metallurgical problems, work will concentrate on corrosion-fatigue as funded by EPRI.

job	file	date	report no.
84-875-80-200	823.1	March 4, 1987	87-36-K



ontario hydro research division

EXECUTIVE SUMMARY

MATERIALS INTEGRITY SECTION: A REVIEW OF 1986 AND PROPOSALS FOR 1987

R.G. Fleck

The Materials Integrity Section performs metallurgical studies in support of the nuclear and thermal generating plants. The main emphasis is to develop an expertise to assess materials problems in operating plant and to develop predictive capabilities to determine when extensive rehabilitation is required. The work programs are mainly concerned with pressure tubes where the emphasis is on: the impact of hydrides on the integrity of the tubes and the determination of the metallurgical and operating parameters which control the in-reactor deformation.

In the area of hydrides the parameters which influence crack initiation and crack growth are being studied. A direct relationship between the tensile strength of the pressure tube and the crack growth velocity has been established. Also in the hydride studies, a program to develop a technique to nondestructively measure the hydrogen concentration was started. The initial results are encouraging. The deformation studies were concentrated on the analysis of in-service measurements of elongation, diametral and sag and on the continuation of the irradiation growth program in the Dido reactor. Some specimens in the irradiation growth study have exceeded the neutron fluence of pressure tubes in the lead reactors. This will provide information to predict pressure tube deformation in the reactors.

The installation of some TG3 tubes in Bruce 8 is significant and over the next few years in-service deformation of these tubes will be monitored. For future LSFCR, the Task Group 3 Route 1 tubes may be used if their predicted reduced elongation rates are confirmed by the Bruce 8 measurements.

In support of the hydride and deformation studies detailed transmission electron microscopy is performed. This provides information on the microstructural parameters of pressure tubes which could explain some of the observed in-service behaviour.

Studies of fuel clad material have concentrated in the assessment of the mechanical properties of oxidized material. This information is required to establish "handling" criteria for fuel bundles that for a variety of reasons may be exposed for prolonged periods to air.

In support of thermal plants, the main thrust of the work will be in corrosion-fatigue as part of an EPRI funded project. The test facility has been constructed and preliminary tests to commission the equipment and test technique will be performed in the near future.

[Handwritten signature]
[Handwritten initials]

job	file	date	report no.
84-875-80-200	823.1	March 4, 1987	87-36-K

TABLE OF CONTENTS

	<u>Page</u>
1.0 INTRODUCTION	1
2.0 MATERIALS INTEGRITY SECTION PERSONNEL	2
3.0 1986 REVIEW	2
3.1 In-Reactor Deformation	2
3.1.1 Analysis of Power Reactor Dimensional Changes	2
3.1.2 Tube Elongation	2
3.1.3 Diametral Expansion	2
3.1.4 Tube Sag	3
3.1.5 Irradiation Growth	3
3.1.6 Irradiation Creep	4
3.1.7 Out-Reactor Deformation	4
3.2 Hydrides in Pressure Tubes	4
3.2.1 TRAC Program	4
3.2.2 Assessment of Blister Growth in Reactor	5
3.2.3 Crack Initiation in Pressure Tube Materials	5
3.2.4 Rolled Joint Integrity	6
3.2.5 Load Following	7
3.2.6 Hydrogen Diffusion in P8 Fretted Tubes	7
3.2.7 Delayed Hydride Crack (DHC) Growth	7
3.2.8 Other DHC Growth Studies	8
3.2.9 Pressure Tube Offcuts Examination	9
3.2.10 Effects of Hydride Morphology on Fracture toughness of Zr-2.5 Nb Pressure Tubes	9
3.2.11 Crack Initiation from Pickering 8 Fretted Tubes	9
3.2.12 Fuel Bundle Scratch Test	10
3.2.13 Detection of Hydrides in Pressure Tubes	10
3.2.14 Committee Participation	10
3.3 Transmission Electron Microscopy Studies of Zirconium Alloys	11
3.3.1 Dislocation Substructure in Zirconium Alloys	11
3.3.2 Effects of Internal Stresses on Hydride Precipitation in Zircaloy	12
3.3.3 Stability of the Zr-2.5 Wt% Nb Pressure Tube Microstructure	13
3.4 Materials Properties for CANDU Reactor Core-Component Alloys	13
3.4.1 Fundamental Studies of Elevated Temperature Deformation Behaviour	13
3.4.2 Biaxial Deformation of Zr-2.5 Wt% Nb Pressure Tube Samples	14
3.4.3 Materials Properties of CANDU Reactor Zircaloy-4 Fuel Cladding	14
3.4.4 High Temperature Deformation Laboratory	16
3.4.5 Materials Properties for CANDU Reactor Ex-Core Component Alloys	16
3.5 Pressure Tube Development	18
3.6 Corrosion-Fatigue	
3.6.1 EPRI Project	18
3.6.2 Corrosion Fatigue-Crack Propagation	19
3.6.3 Corrosion-Fatigue of Carbon Steel Components in Generating Units	19
3.6.4 Corrosion-Fatigue of Pressure Tube Material	19

TABLE OF CONTENTS (Cont'd)

	<u>Page</u>
3.7 Thermal Plant Life Assessment	19
3.8 Boiler Tube Failure Investigations	20
4.0 1987 PROGRAMS	20
4.1 In-Reactor Deformation	20
4.1.1 Analysis of Power Reactor Dimensional Changes	20
4.1.2 Irradiation Growth	20
4.1.3 Irradiation Creep	20
4.2 Out-Reactor Deformation	20
4.2.1 Study of Anisotropic Creep	20
4.3 Hydrides in Pressure Tubes	21
4.3.1 TRAC Program	21
4.3.2 Reactor Assessment Studies	21
4.3.3 Crack Initiation in Pressure Tube Materials	21
4.3.4 Rolled Joint Integrity	21
4.3.5 Load Following	22
4.3.6 P8 Fretted Tube Hydrogen Diffusion	22
4.3.7 DHC Growth Studies	22
4.3.8 Fracture Toughness Evaluation	22
4.3.9 Crack Initiation from Pickering 8 Tubes Containing Fretted Marks	22
4.3.10 Detection of Hydrides in Pressure Tubes	23
4.3.11 Committee Participation	23
4.4 Transmission Electron Microscopy	23
4.4.1 Dislocation Substructures in Zr-2.5 Wt% Nb Pressure Tube Alloys	23
4.4.2 Hydride Precipitation	23
4.4.3 TEM Studies of Irradiated Zr-2.5 Wt% Nb Pressure Tubes	24
4.4.4 Dislocation Substructure in Zircaloy Pressure Tube Alloy	24
4.4.5 TEM Investigation of the Bainite Transformations in Fe-Ni-C, Fe-Si-C and Fe-Si-Mo Alloys	24
4.5 Materials Properties for CANDU Reactor Core-Component Alloys	24
4.5.1 Fundamental Studies of Elevated Temperature Deformation Behaviour	24
4.5.2 Materials Properties for Zircaloy-4 Fuel Cladding	25
4.6 Materials Properties for CANDU Reactor Ex-Core Component Alloys	25
4.6.1 Darlington PHT Piping Material	25
4.6.2 General Materials Testing	25
4.7 Pressure Tube Development	25
4.7.1 Pilgered Pressure Tubes	25
4.8 Corrosion-Fatigue	26
4.8.1 Corrosion Fatigue-Crack Propagation	26
4.8.2 Corrosion Fatigue-Crack Initiation	26

TABLE OF CONTENTS (Cont'd)

	<u>Page</u>
5.0 SUMMARY	26
REPORTS	28
PUBLICATIONS	31
PRESENTATIONS	32
CONFERENCES AND TECHNICAL MEETINGS	33
DISTRIBUTION	Last Page



**ontario hydro
research division**

TO: D. Mills
Director of Research

**MATERIALS INTEGRITY SECTION
A REVIEW OF 1986 AND PROPOSALS FOR 1987**

1.0 INTRODUCTION

Throughout 1986 extensive effort was put into the upgrading of the test capabilities. There is now available a Tube Test Facility which is capable of performing burst tests and long term creep tests on a variety of tubular components. Another facility which was developed in 1986 is a test loop with controlled water chemistry which, in conjunction with a hydraulic test frame, will be used to determine the parameters which control the initiation of corrosion-fatigue damage in thermal plant boiler tube materials. Currently under construction is a second corrosion-fatigue facility to test other materials including pressure tubes. The Materials Characterization Test Facility was upgraded with the purchase of a high resolution laser strain gauge which will enable the micro yield of materials to be evaluated which will provide more detailed information for use in computer codes to evaluate flaws and material integrity.

The main emphasis of the Materials Integrity Section's programs is in the areas of pressure tube cracking and deformation. These studies are part of the joint OH-AECL Fuel Channel Research Program. The general thrust of the programs is to develop a realistic predictive capability so that reactor rehabilitation programs can be effectively planned. At present such decisions are largely based on extrapolation of in-service data. An important aspect of the deformation studies is that some specimens in the Dido irradiation growth program have now exceeded the fast neutron fluence of pressure tubes in our lead reactors. These types of experiments will now be able to identify some of the materials behaviour at high fluences before the reactors.

For support of thermal plant metallurgical problems, the emphasis was on residual life assessment. As a result of the Lakeview life assessment program which is dependent on the evaluation of metallurgical replicas H.J. Westwood transferred to the Metallurgy Section for the duration of these life assessment studies. This has resulted in a decreased support for thermal plants from the Materials Integrity Section where the efforts will be concentrated on the corrosion-fatigue program sponsored by EPRI. The thermal plant metallurgy programs also suffered because of the departure of J.D. Parker.

2.0 MATERIALS INTEGRITY SECTION PERSONNEL

R.G. Fleck
L.J. Compton

Supervising Engineer
Secretary

Engineering Staff

E.T.C. Ho
W.K. Lee
M. Leger
V. Perovic
G.K. Shek
J.M. Tadros
K.M. Tashiro
H.J. Westwood (8 mos)
J. Parker (9 mos)

Technical Staff

A. Donner
Z. Grozdanovski
R.E. Irwin
R. Jarochowicz
O. Kiir
M.J. McGraw
D.A. Oad
H. Seahra
P.J. Vesely

Attached Staff

Consultant Dr. D.V. Leemans
(Cross Currents Research & Policy
Consulting Ltd)

Summer Students

Mr. G. Storus
Mr. J. D'Angelo
Miss S. Unterbergs

3.0 1986 REVIEW

3.1 In-Reactor Deformation

3.1.1 Analysis of Power Reactor Dimensional Changes

Service operation of CANDU reactors has shown that deformation of pressure tubes occurs with time. Three different types of dimensional changes are measured periodically; namely, tube elongation, diametral expansion and tube sag. Analysis of these dimensional changes in terms of reactor operating conditions and tube metallurgical parameters has been carried out for selected tubes in Pickering NGS-A Unit 3 and Bruce NGS-A Unit 2. Progress may be summarized as follows:

3.1.2 Tube Elongation

Variation in pressure tube elongation with fluence appears to be due to real variations in in-reactor performance of individual tubes. This behaviour does not appear to be a simple consequence of changes in operating conditions. Furthermore, no rationalization could be made on the basis of texture. Estimates of the observed elongation behaviour could be obtained using the AECL design equation (CRNL 1993) and average tube operating conditions and metallurgical parameters. However, when predictions were made for individual tubes using actual parameters, the accuracy of estimates decreased.

3.1.3 Diametral Expansion

Diametral deformation of pressure tubes is a complex function of stress, temperature, and fast neutron flux. A range of deformations was observed for the selection of tubes which does not appear to be related simply to differences in operating conditions. No simple relationship could be established with peak deformation not always coinciding with the location of highest temperature and flux. An indication that diametral strain rate accelerates after ~80 000 EFPH was detected, although additional data is needed to verify this. At this time, accumulated hoop strain does not create a concern with respect to creep rupture or reduction in critical channel power and flux limits. Estimates of the observed diametral expansion behaviour made using the

AECL design equation (CRNL 1993) and average tube operating conditions and metallurgical parameters were poor for the Pickering channels and reasonable for the Bruce channels. When actual tube parameters were substituted into the design equation, the estimates were significantly worse.

3.1.4 Tube Sag

Preliminary analysis of plant sag data has revealed that a range of deflections exists which cannot be explained simply in terms of operating conditions. Maximum deflection was not always coincident with the location of highest temperature and flux. Estimates of the magnitude and location of the maximum sag made using the Ontario Hydro computer code CDEPTH (which uses the AECL design equation CRNL-1993) were reasonable for some tubes but extremely poor for others. Furthermore, the poor agreement between the measured pressure tube deflection profile and the profile predicted using CDEPTH is further evidence of the need to improve the predictive capability of the computer code.

In general, the current design approach (ie, the AECL design equation CRNL-1993) does not provide the predictive capability necessary to plan long-term operating and maintenance schedules. Further experimental work is required to identify the factors affecting in-reactor deformation and to add to the database of information which can be used to develop a model of power reactor performance from laboratory studies. Plant dimensional measurements could then be used to validate the model.

3.1.5 Irradiation Growth

Analysis of the latest information from the Zr-2.5 wt% Nb irradiation growth program being carried out in the DIDO reactor at AERE Harwell has been completed. The findings may be summarized as follows:

1. Longitudinal samples irradiated at 280°C continue to grow at an approximately steady growth rate at fast ($E > 1\text{ MeV}$) neutron fluences of $\sim 6 \times 10^{25} \text{ n}\cdot\text{m}^{-2}$. Transverse samples irradiated at 280°C continue to contract at an approximately steady contraction rate at fast neutron fluences of $\sim 6 \times 10^{25} \text{ n}\cdot\text{m}^{-2}$.
2. Previously unirradiated longitudinal samples irradiated at 310°C and 350°C show an initial positive offset followed by contraction at higher fluences. The magnitude of the initial positive offset at 310°C is lower than that observed at 280°C, and the positive offset at 350°C is lower than that at both 280°C and 310°C. Longer term specimen contraction was observed at both 310°C and 350°C than for samples which exhibited positive growth at 280°C.
3. Increasing the irradiation temperature to 310°C or 350°C after a prior irradiation at 280°C produces a contraction in material cut longitudinally to the pressure tube axis. The greater contraction occurs at the higher temperature. The magnitude of the contraction does not appear to be related to the accumulated fluence.
4. Transverse material previously irradiated at 280°C shows an initial positive offset and then remains approximately constant when irradiated at 310°C.

The implications of these results to the service performance of pressure tubes are presently under consideration. It appears that material cut from the same tube can exhibit markedly different growth behaviour over a relatively small temperature range. Even if this behaviour is a transient effect and long-term positive growth is eventually achieved at all temperatures, the extent of the observed contraction may have important implications for pressure tubes operating at temperatures above 300°C (ie Darlington tubes will have outlet temperatures of ~313°C). Furthermore, the existence of strain transients for small changes in temperature may have important implications to cyclic operation and possibly to improving tube life (by in-situ annealing).

3.1.6 Irradiation Creep

Small tubes machined from Zr-2.5 wt% Nb pressure tube walls are presently being irradiated at 280°C in the DIDO reactor at AERE Harwell. The information from these tests will add to the small data base that is now available on in-reactor creep of pressure tubes and also provide a scoping study leading to the design of more detailed experiments.

3.1.7 Out-Reactor Deformation

Study of Anisotropic Creep

Because of the hexagonal close packed crystal structure of zirconium, processes such as irradiation creep and growth are highly anisotropic. For the design of reactors, it is important to predict the anisotropic deformation of the various components (eg, pressure tubes, calandria tubes, fuel cladding) throughout the reactor life. The purpose of this study is to develop a procedure to predict anisotropic deformation from compression creep tests. The progress of this study may be summarized as follows:

1. The compression cage to be used to carry out the experiments was designed and constructed. The cage was designed to screw into the pull rods of a conventional tensile creep machine.
2. Preliminary tests carried out on cylindrical slugs of 2-1/4 Cr-1 Mo steel revealed a number of problems. It was found that it is extremely difficult to achieve the alignment necessary to give axial loading of the specimen. Also, friction at the specimen/cage interface led to barreling of the test sample.

Presently, data from the preliminary tests are being analyzed and possible solutions to the experimental problems are being considered.

3.2 Hydrides in Pressure Tubes

3.2.1 TRAC Program

Work on the TRAC program continued throughout the year, although progress on the blister growth and fracture was somewhat slower than had been anticipated at the beginning of the year. There were several factors which limited the output, among them being the need for the key technologist to spend a considerable effort on a program which was deemed to be of immediately higher priority (P-8 fretted tube diffusion); malfunction of the capacitance gauge measurement equipment in detection of blister growth; and finally considerable pressure on M. Leger in terms of other work which is described below.

Administration of the program is being taken over by NSD and this also led to significant demands for reporting of financial status etc, which was not accessible to them. Changes in the administration process are required for continuation in 1987.

Despite the level of work being lower than anticipated, there were some significant achievements in the growth and fracture area which resulted from work carried out earlier. In particular, the measurements of the deuterium profiles through deuterided blisters grown with the air jet technique were shown to be in excellent agreement with the model predictions. Deuteride blisters grown under thermal cycling conditions were also completed and sent to Queen's University for analysis. The Queen's work has also recently shown that the deuterium in a blister at the initiation site of the G-16 failure has substantially the same distribution as some of the blisters grown in the laboratory. Blisters were successfully grown under stress and have been fracture tested by the Nondestructive and Fracture Evaluation Section. They are awaiting metallography.

Technical support was provided to the oil loop test of blister growth, a test which is designed to provide insights into the effect of blister growth on the boundary condition at the contact between pressure tube and calandria tube. The instrumented tube was successfully hydrided.

3.2.2 Assessment of Blister Growth in Reactor

This work became a major program this year. It was carried out in cooperation with Dr. D.V. Leemans of Cross Currents Research and Policy Consulting Limited who designed and carried out the Monte Carlo studies. The ultimate objective of this work is the prediction of the probability of pressure tube failures in particular reactors as a function of time. So far the model has been developed only to the point of prediction of the probable number of tubes having blisters.

There were two major studies done using this model during the year: the first was for those reactors being considered for the SLAR operation; the second was for the technical assessment of the REFAB option. These studies are already having considerable impact on the decisions of tube removal in the coming years and have been used by AECL-CO to make a case for a staged SLAR operation for the CANDU-600 units.

The studies have demonstrated the sensitivity of the assessment of the probable numbers of blistered tubes to many different parameters and point to the research areas which are most likely to be of benefit in reducing the uncertainty in the assessment. In particular, the deuterium ingress rate and the threshold deuterium level for blister formation are very influential in determining the outcome.

3.2.3 Crack Initiation in Pressure Tube Materials

The main objectives of this project were to generate engineering data and to interpret existing data on crack nucleation. A better understanding of the factors leading to crack initiation will be the final product. The test matrix included variation in the applied stress, hydrogen content, surface condition and thermal cycle.

The in-house development of the four point bend test procedures was successful, and crack nucleation was observed in the validation experiment. The role of stress relaxation was not significant. Any relaxation that had occurred did not affect crack initiation in the deflection

controlled four point bend tests. The specimen with 60 ppm hydrogen was loaded to 500 MPa under thermal cycling condition, and the dc potential drop technique was used to monitor the 0.25 mm deep 45 degree angle notch running from edge to edge along the axial direction of the specimen. After 29 thermal cycles with peak temperature of 300°C and holding at 160°C, crack extensions measured on the fracture surface were found to correspond to the incremental changes in the crack voltages.

The monitoring of the test temperatures and the potential drop voltages in notched specimens was accomplished by the use of a multitasking data logger. The computer program using the C language was developed on a multitasking single user UNIX microcomputer interfaced to a precision analog-to-digital data recorder. This system allowed multi-execution of the data logging program to record the experimental data for each specimen in the test matrix. This has resulted in better logistics and greater flexibility in the start and stop of any test.

To simulate the continuous hydrogen ingress occurring in the fuel channel, the temperature gradient approach was chosen for the crack initiation experiments. A four point bend specimen with a hydride layer on the OD surface was heated at the two ends by small heaters, and the whole specimen was placed inside a furnace. The higher temperature at the two ends generated the energy gradient to drive the hydrogen from the surface hydride layer to the center of the specimen. With the center held at 250°C and a temperature gradient of 35°C a one dimensional diffusion model predicted that the first new hydride in the specimen would be formed in about 6 months for a specimen containing 30 ppm hydrogen. If the specimen contained 60 ppm hydrogen, the time required would be about one month. Specimens with or without notches were loaded up with this temperature gradient setup, and to-date no cracking has been observed. Further improvement in the magnitude of the temperature gradient was also carried out. A 50°C temperature difference was achieved using a new setup procedure.

An initial attempt to interpret all the existing cantilever beam data was completed, the exponential life function was found to provide a reasonable description of the data. This function has been used extensively for the determination of the life distribution of electronic components which individually can see many different type of service cycles. For the pressure tubes in a reactor, there are variations in the stress level and the type of defects and the exponential life function may represent the material behaviour in such environment. The statistical significance test showed that the data base on crack initiation was in agreement with the exponential life distribution. Further development is required to establish the correlation between the crack initiation life distribution and other controlling variables such as stress intensity and hydrogen level.

3.2.4 Rolled Joint Integrity

The zero clearance rolled joint was developed to eliminate the high residual stresses present in early type of Candu fuel channels, this program was designed to investigate the rolled joint's performance at high hydrogen level. To simulate the service condition, production rolled joints with surface hydride layers were prepared for inert gas pressurization at operation temperature.

Two pressure tubes with hydride layers on the OD surfaces were forwarded to Canadian General Electric for installation of the production type of zero clearance rolled joints. The highest allowable clearance and the maximum permissible misalignment were chosen in the preparation of the hub for the rolling operation. Post rolling profile measurement indicated that the maximum residual hoop stress was of the order of 54 MPa.

3.2.5 Load Following

The effect of this change in operation on crack initiation is largely unknown, a survey of the operation parameters and the design of a scoping test were therefore carried out. During load following operation, the pump heat generated within the primary heat transport system can maintain the inlet temperature while the outlet temperature can be down to about 255°C at 0% power output. The maximum change at the outlet temperature end will be about 50°C during a load following maneuver. The system pressure is not expected to vary much. In parallel with this increased load following, there may be more start up and shutdown of the reactor due to excess total baseload capacity. The test conditions used in laboratory experiments must include these changes as well as the worst case situation. For the initial tests, the tensile type of specimens will be thermally and mechanically cycled to determine the worst possible material damage.

3.2.6 Hydrogen Diffusion in P8 Fretted Tubes

To determine if the hydrogen concentrated at the fretted area will diffuse away under reactor operating conditions, fretted tubes were assembled for inert gas pressurization to 10 MPa at 250°C. After long hold time at the operating pressure and temperature, the hydrogen distribution was determined by optical metallography and hydrogen analysis.

For the diffusion experiments, two short pieces of fretted tubes were assembled using electron beam welding. The electron beam welding was developed to join an extension to the short fretted piece to allow room for hydrogen redistribution. Since the as received tubes did not contain the pit-like secondary marks, a high speed microdrilling technique was used to put in pits with stress concentration factor of 2.5 onto the ID surface at the fretted region.

After 63 days of diffusion anneal at 250°C while under an internal inert gas pressure of 10 MPa, one fretted tube with simulated secondary marks was removed for metallographic examination. Reoriented hydrides were generally found under the fret mark. However, no microcracks were discovered under any of the secondary marks examined. Semiquantitative measurement of the hydride apparent volume fraction indicated that hydrogen redistribution had occurred, and the profile was similar to that predicted by a theoretical model based on the receding hydride/nonhydride interface equation. Hydrogen analysis of an axial radial section showed a hydrogen redistribution profile higher than that of the model prediction. The nonuniform initial hydrogen distribution may have accounted for some of these differences.

The diffusion of hydrogen away from the fret mark was therefore confirmed in the first tube removed from the furnace. There was no apparent tendency for the secondary marks to prevent the hydrogen from diffusing away and no crack initiation had occurred at the secondary marks.

3.2.7 Delayed Hydride Crack (DHC) Growth

The objectives of the program are: (1) to evaluate the effects of hydride morphology, thermal history and pressure tube variability on delayed hydride crack velocity in Zr-2.5 Nb pressure tubes; (2) to provide more crack growth data on pressure tubes at reactor operating temperature. Progress in 1986 includes the following:

1. The computerized data acquisition system was upgraded with a disk drive and a plotter for data storage and analysis.

2. DHC velocity in pressure tubes is strength dependent. Tubes with higher strength have higher DHC velocity. The variation in strength was achieved either by stress relief or by tube fabrication procedure. Current cold worked pressure tube materials were stress relieved at 500°C, 550°C and 600°C to lower the strength. Samples from the TG3 hold points tubes and the thick-end replacement tubes were chosen to give different levels of strength. The hardnesses of the samples were measured and the specimens were tested at 250°C. The results showed that there was a linear relationship between crack velocity and hardness. The fracture surfaces showed that the striation spacing increases as the hardness decreases. The size of the striation spacing corresponded to that of the reoriented hydride zone at the crack tip. The crack advanced in steps of the size of the striations.

It is postulated that materials with lower strength require a larger embrittled zone for the crack to propagate.

3. There might be a critical requirement of strength for DHC to occur. The samples that were stress relieved at 600°C could not be cracked until the test temperature was lowered from 250°C to 230°C. The lowering of the test temperature increases the matrix strength.

Another tests showed that a regular pressure tube sample containing 60 ppm hydrogen could not be cracked at 290°C under an applied K of 12 MPa $\sqrt{m}$. The test temperature was decreased in steps of 5°C until cracking finally occurred at 270°C. A quenching experiment showed that hydrides were present at 290°C. Hence the absence of cracking at 290°C might be due to the low matrix strength.

4. DHC velocity was higher and K_{1H} may be lower during cooling than under isothermal conditions.

3.2.8 Other DHC Growth Studies

Several DHC tests were performed for CNS with specific loading and thermal histories to simulate reactor operating conditions.

1. **Effectiveness of temperature maneuver in slowing DHC** - A test was performed which showed that approaching reactor shutdown temperature at 180°C from below did not result in drastic reduction in crack velocity.
2. **Bruce Unit 2 N06 DHC test** - A DHC growth test was performed with a similar thermal history as Bruce Unit 2 N06 after the reactor was poisoned out and before the annulus became saturated. The amount of crack extension and the periods in which cracking occurred were determined.
3. **Depressurization** - A test was performed to examine the effects of K reduction simulating reactor depressurizing on DHC behaviour after a temperature maneuver to slow down crack growth. The results showed that an incubation period was required for DHC to resume after K reduction. The results also raised some concern about the effectiveness of temperature maneuver in slowing/arresting crack growth.

3.2.9 Pressure Tube Offcuts Examination

Bruce Unit 2 - The failure of Bruce Unit 2 channel NO6 in March 1986 was determined to be caused by a manufacturing defect at the outlet end of the pressure tube. Manufacturing defects in pressure tubes also led to the leakage of channel J11 in 1982 and the pre-service replacement of channel K02 in Bruce Unit 2. The defects in channels NO6, J11 and K02 all extended to the end of the pressure tubes and were subsequently observed in the tube offcuts by ultrasonic and metallographic examination.

A program to examine all Bruce Unit 2 pressure tube offcuts for manufacturing defects was undertaken. The offcuts were ultrasonic inspected and examined metallographically. For metallographic examination, the inboard face of the offcuts was polished and about a quarter of the wall thickness was examined along the inner surface at 100X magnification. All available Bruce Unit 2 offcuts were examined and no significant defects were found.

Zirconium Carbide Inclusions - High frequency ultrasonic inspection detected indications in offcuts of a batch of Pickering Unit 5 pressure tubes from one ingot. Metallography showed that the offcuts contained cracked inclusions which were determined to be zirconium carbide by Auger analysis.

3.2.10 Effects of Hydride Morphology on Fracture Toughness of Zr-2.5 Nb Pressure Tubes

This is a co-operative program with the Nondestructive and Fracture Evaluation Section. Earlier tests using compact tension specimens machined from straightened strips showed that the presence of radial hydrides or "mixed" hydrides significantly reduced the low temperature fracture toughness of Zr-2.5 Nb pressure tubes (OHRD Report 86-93-K). However, the specimens did not have the mixed hydride morphology observed in pressure tubes removed from reactors due to residual stresses introduced by flattening the material.

The main objective of this test program is to determine the fracture toughness of Zr-2.5 Nb pressure tube material containing a mixed hydride morphology similar to that observed in some pressure tubes removed from reactors. This requires hydriding a tube section under internal pressure and performing the fracture toughness tests using small curved compact tension specimens.

A welding procedure was developed to join a titanium end-plug to pressure tube by electron beam welding. A tube section was hydrided to 60 ppm hydrogen under an internal pressure of 12.4 MPa (equivalent to 160 MPa hoop stress). The hydrides obtained were predominantly "mixed" and oriented between 40°-80° from the transverse direction and similar to that observed in some pressure tubes removed from reactors. Small curved compact tension specimens will be machined from this tube to determine the fracture toughness as a function of temperature.

3.2.11 Crack Initiation from Pickering 8 Fretted Tubes

Pit-like defects were put into the inner surface of three hydrided four-point bend specimens by high speed microdrilling to simulate the secondary fretted marks observed in the P-8 tubes. The drilled defects had different geometry (round, square and V-shape) and varied in depths from 0.2 to 1.0 mm. This represented estimated stress concentration factors of about 1.5 to 3.0. The samples were subjected to five thermal cycles under an applied stress of 260 MPa. After the

thermal cycling, an almost through wall crack was developed from the 1.0 mm deep V-shape defect and a 0.5 mm deep crack was initiated from the 0.8 mm V-shape defect. The cracking mechanism was determined to be delayed hydride cracking. No cracking was detected in the other defects, but all of them including the 0.2 mm deep round defect contained radial hydrides at the root.

3.2.12 Fuel Bundle Scratch Test

The objective of this experiment is to determine if fuel bundle scratches can initiate delayed hydride cracking in Zr-2.5 Nb pressure tubes containing radial hydrides at the inner surface. Specimens containing radial hydrides were scratched by a loaded stylus and then thermally cycled under applied stresses up to 500 MPa in four-point loading jigs. No cracking was observed associated with the simulated fuel bundle scratches. The results indicated that fuel bundle scratches are not likely to initiate DHC even with the presence of radial hydrides in the pressure tubes.

3.2.13 Detection of Hydrides in Pressure Tubes

The major goal of 1986 was the identification of a technique for the nondestructive evaluation of hydrogen in Zr-2.5 wt% Nb pressure tube material. An extensive literature survey was undertaken and it was determined that resistivity and thermoelectric power would be the physical properties that would be investigated first. The technique relies on detecting a discontinuity in the temperature dependence of these physical properties due to hydride dissolution and precipitation. Thus, the terminal solid solubility (TSS) temperatures are measured and the total hydrogen concentration are determined from this data. A limited amount of data was available for the effects of hydrogen on resistivity in Zr - 2.5 wt% Nb pressure tube material, however similar data for thermoelectric effects of hydrogen was not available. A preliminary measurement for the effects of hydrogen on the thermoelectric power of Zr-2.5 wt% Nb was performed. The data demonstrated strong discontinuities due to hydride dissolution and precipitation. The construction of a small sample experimental setup was initiated in order to improve accuracy and reproducibility in measurements for both resistivity and thermoelectric effects.

The determination of hydrogen concentration requires a good correlation between concentration and TSS temperatures for both irradiated and nonirradiated material. The technique used to establish this correlation must be able to perform this with good reproducibility, accuracy and ease of measurement (especially for irradiated material). Differential calorimetry was investigated and found to be able to measure TSS for samples with as little as 14 ppm hydrogen with a sample size as small as 100 mg.

3.2.14 Committee Participation

During the past year, input was provided to three different committees: An NSD SLAR assessment committee; The Bruce N06 Failure Analysis Group; and, the REFAB Technical Assessment Committee. Of these three, the greatest level of work was required by the SLAR assessment group which produced a technical assessment of the need for SLAR for Bruce Units 3 and 4 and Pickering Units 5 and 6.

3.3 Transmission Electron Microscopy Studies of Zirconium Alloys

Transmission electron microscopy was very extensively employed during 1986 in studying metallurgical problems related to the integrity of components in nuclear plants. The research is a further continuation of the current programs whose main objective is to gain general and fundamental knowledge of the microstructure and its relationship to the physical properties of the materials used in nuclear plants. These studies are basically divided into two groups:

- i) The first group is related to the characterization of dislocation substructure and phase transformation in Zr-2.5 wt% Nb and Zircaloy-2 alloys subjected to various thermo-mechanical treatments.

The main purpose of these investigations is to obtain more fundamental information on the role of dislocation substructure on the mechanical properties and irradiation growth of the pressure tube materials.

- ii) The second program is an attempt to study the role of internal stresses on hydride precipitation in zirconium alloys. The aim of these studies is to further attempt to understand some of the more fundamental questions concerned with hydride precipitation.

Both studies are well underway and some of the results are already published or presented in the open literature.

3.3.1 Dislocation Substructure in Zirconium Alloys

As reported previously, one of the characteristic dislocation substructures of Zr-2.5 wt% Nb pressure tubes is established during extrusion at temperatures that lie in the two phase α - β region of the phase diagram. The neighbouring α -grains are separated by a substructure that contains both $\langle a \rangle$ and $\langle c \rangle$ -component dislocations. When the subboundary lies on a prism plane, both the $\langle a \rangle$ and $\langle c \rangle$ dislocations are in a screw orientation and a pure twist boundary results. As the boundary plane rotates from the prism plane, the $\langle c \rangle$ -component dislocations become mixed and a second set of $\langle a \rangle$ dislocations is observed.

The second characteristic dislocation substructure is established during cold-working and recovery. This substructure is superimposed on the basic substructure established after extrusion. Cold-working at room temperature primarily introduces $\langle a \rangle$ dislocation, but a few $\langle c \rangle$ dislocations are formed. The extent to which the recovery process proceeds depends on the annealing temperature.

The orientation relationship established between the α and β -grains is semi-coherent containing regular substructure. In order to obtain more insight into the nature of this interface substructure, considerable effort during the past year was placed on more quantitative analysis of the interface structure associated with b.c.c. (β)_{Zr} $\rightarrow$ h.c.p. (α)_{Zr} phase transformation. Two important defects are clearly observed at the interface. The first defect is a parallel set of dislocations lying in the interface and the second defect is an array of interface steps. Analysis has been attempted of the interface dislocations in terms of a model based on the invariant line concept, which seeks the conditions of best directional matching between precipitate and matrix lattices. This is the geometrical condition of an invariant (unextended and unrotated) line in the interface as it is used in martensite theory. The transformation stresses which build-up whenever the precipitate grows normal to its zero strain axis can be relieved by the periodic nucleation and

expansion of a shear loop. The loop plane must contain both the Burgers vector and the invariant line and will generally be inclined to the habit plane. As a result, the precipitate plate will be enveloped by a parallel set of shear loops lying in the interface. Because of the plate shape of the precipitate, the appearance of the shear loops in most of the orientation will be one of dipoles along the invariant line. Our T.E.M. examinations have clearly revealed that the observed interface dislocations lie close to the predicted invariant line direction associated with $\beta_{Zr} \rightarrow \alpha_{Zr}$ (h.c.p.) phase transformation.

The second defect observed at the interface is a ledge structure. Although pure steps or ledges without a strain field can be present at interfaces, at the majority of the steps a strain field was observed. This strain field may arise from an array of dislocations or could be associated with the residual elastic strains required to bring the β_{Zr} and α_{Zr} lattices into high planar coincidence configuration. Diffraction contrast of the α/β interface dislocations is very complicated and as a result exact visibility criteria is difficult to establish. The contrast experiments have shown that the dislocations in the interface are out of contrast with (0110) reflection operating and in contrast with (0002), (1120) and (0111) reflections operating. Such contrast behaviour is only consistent with $1/2$ (2113) Burgers vector since the observations of contrast with the (0002) and (1120) reflections operating eliminate $1/3$ $\langle 1120 \rangle$ and $\langle 0001 \rangle$ dislocations, respectively.

It is also clearly established that the α/β interface substructure is very closely linked to the α/α -subboundaries developed after extrusion. These subboundaries contain $\langle a+c \rangle$ dislocations and our understanding of the $\beta_{Zr} \rightarrow \alpha_{Zr}$ phase transformation is essential in order to better understand material behaviour under different conditions. This dislocation substructure will influence the properties of the pressure tube material in two important ways: firstly, by their effect on mechanical properties and secondly through their influence on irradiation growth response of the alloy. The presence of different types of dislocations and their spatial distribution should influence irradiation growth by altering the manner in which irradiation produced point defects are partitioned among sinks.

The preliminary T.E.M. characterization of dislocation substructure in Zircaloy pressure tube has started. The object of this study is to investigate the mechanisms responsible for the rather unusual work-hardening behaviour at relatively low strains in tensile testing of transverse Zircaloy-2 pressure tube samples at room temperature. The first preliminary results indicate that this behaviour could be associated with microtwinning. This examination is part of a study to improve the understanding of the deformation of zirconium.

3.3.2 Effects of Internal Stresses on Hydride Precipitation in Zircaloy

The study of the role of internal stresses on hydride precipitation in Zircaloy-2 swaged rod continued. It is shown that the distribution of hydride precipitates in annealed and/or cold-worked Zircaloy is determined in part by the residual stresses left in the material after cooling from high temperature or cold-working. The origin of the internal stresses lies in the "grain interaction stresses" associated with high thermal and/or plastic anisotropy of zirconium. Recent calculations and neutron diffraction measurements have shown that the residual stresses are sensitive both to the magnitude of the imposed plastic strain and grain orientation. The calculations and neutron diffraction experiments are done by S.R. MacEwen at AECL-CRNL.

The T.E.M. results support the viewpoint that the pattern of hydride precipitation varies from grain to grain according to the distribution of "grain interaction stresses".

In the tensile specimens (+4% ϵ) hydrides grow either along the α/α boundaries or transgranularly. The transgranular hydrides have commonly observed habit plane close to the (0002)_{Zr} planes, but where these transgranular hydrides are observed, the crystallographic orientations of the grain (c-direction) lie at an acute angle to the tensile axis. This indicates that the observed transgranular hydrides grow in particular grains in which the (0002) planes are under tension, as expected according to the calculations and neutron diffraction measurements. In the compressive specimens (-4% ϵ) hydrides grow along the interfaces only, either along the α/α boundaries or twin interfaces. The stress state in the compressive specimens when the major mode of deformation is twinning is not known at present. However, the fact that only interface hydrides are observed indicates that twinning has relieved most of the residual tensile stresses in grains. In the annealed samples (650°C) regardless of the previous mechanical history (tensile or compressive deformation) hydrides grow mainly transgranularly as predicted by the calculations and neutron diffraction measurements. The residual stresses acting on the basal plane after cooling from high temperature are tensile. This tensile stress would help to accommodate the volume misfit between hydride and α -Zr lattice in a direction normal to the habit plane, which is close to the (0001)_{Zr}. This study is a part of the joint program between Ontario Hydro (V. Perovic, M. Leger), AECL-CRNL (S.R. MacEwen) and University of Toronto (G.C. Weatherly).

3.3.3 Stability of the Zr-2.5 Wt% Nb Pressure Tube Microstructure

A T.E.M. study on the stability of microstructure in the Zr-2.5 wt% Nb pressure tubes as a function of stress relieving temperature was initiated. It is intended to correlate the microstructure with hydride precipitation and the results will be incorporated into the delayed hydride cracking program. The first preliminary results indicate that the initial microstructure becomes progressively unstable by increasing annealing temperature from 500 to 600°C. This instability is local in nature producing very large α -grains as a result of abnormal grain growth. These grains are one order of magnitude larger than the average α -grains in the pressure tube.

3.4 Materials Properties for CANDU Reactor Core-Component Alloys

3.4.1 Fundamental Studies of Elevated Temperature Deformation Behaviour

The elevated temperature deformation behaviour of Zr-2.5 wt% Nb PT material has been studied extensively by workers at AECL-WNRE and at OHRD over the past several years. The experimental programs carried out include both uniaxial and biaxial tests. From detailed analyses of the transverse uniaxial data, Shewfelt et al were able to develop two phenomenological creep correlations which were able to describe the elevated temperature deformation behaviour in a reasonable manner. However, comparisons between WNRE creep correlation predictions and some OHRD biaxial test data indicated that the experimentally observed deformation rates were significantly lower than the calculated rates. Since the WNRE creep correlations were developed using uniaxial data obtained from testing in high-purity argon environments, it was reasoned that the observed differences in deformation rates may be attributed, at least in part, to oxygen-strengthening of the Zr-2.5 Nb PT material in an air environment. In addition, NSSD personnel have been investigating some postulated LOCA scenarios in which the PT temperatures may exceed 1200°C, the maximum temperature for the WNRE creep correlations. Therefore, in view of the above and since no data exist for Zr-2.5 Nb PT material above 1200°C, a small test program was carried out at OHRD to characterize the deformation behaviour in an air environment over the temperature range 1200-1500°C.

Results obtained indicate that for a given temperature in the range studied, the range of applied stresses needed to produce significant deformation rates is quite narrow. At relatively low stresses, the deformation is apparently controlled by the growth of a strong oxygen-stabilized α -Zr layer beneath the oxide phase. The resultant deformation rate is very low until a significant fraction of the substrate β -Zr phase has been oxidized and/or after a certain plastic strain has been surpassed. At sufficiently high stresses, the initial deformation rate is high, and as a result, the sample deforms rapidly to failure. The test results also showed that at 1200°C, the WNRE creep correlation of Shewfelt et al overpredicts the observed deformation rates by about an order of magnitude at the highest applied stresses investigated, and by greater than two orders of magnitude at the lowest applied stresses examined.

3.4.2 Biaxial Deformation of Zr-2.5 Wt% Nb Pressure Tube Samples

Biaxial test results obtained in 1984 at OHRD indicated that an internal differential pressure of 0.25 MPa was insufficient to cause significant further deformation when Zr-2.5 Nb PT samples were heated above about 900°C in an internal steam environment. Detailed analysis of the test data showed that a very large fraction of the diametral expansion occurred over the temperature range 627-900°C when the samples were heated at 3°C/s from about 300°C to the upper hold temperatures. The data showed clearly that the combined effects of oxygen-strengthening, changes in crystal structure and changes in grain morphology had substantially enhanced the PT material's resistance to deformation. However, since only a limited number of tests were carried out, it was not possible to determine what combinations of internal pressure and upper hold temperature were critical in slowing down or significantly suppressing the biaxial expansion. As a result, a series of biaxial tests was performed to determine what the critical combinations of internal pressure and temperature were.

Results obtained indicate that for upper hold temperatures in the range 1000-1137°C, the critical pressure lies in the range 0.25-0.3 MPa. Detailed comparisons of data obtained from testing in argon/argon (ID/OD) environment with those obtained in steam/argon environment indicate that there is additional strengthening of the PT material when the deformation occurs in a steam environment. However, this additional strengthening is not as great as that associated with the transformation to a completely β -Zr phase material. These findings are shown clearly in the plots presented in Figure 1. The results also showed that there were no significant differences in deformation behaviour in steam/argon and steam/CO₂ environments. In addition, the experimental data showed that at the highest temperatures investigated (~1327°C), a differential internal pressure of 0.3 MPa was still too high to suppress significant ballooning of the Zr-2.5 Nb PT samples.

3.4.3 Materials Properties of CANDU Reactor Zircaloy-4 Fuel Cladding

Materials data on the oxidation and deformation characteristics of CANDU reactor Zircaloy-4 (Zr-4) fuel cladding for temperatures in the range 400-900°C are very limited. Unfortunately, such data are needed in the analysis of a number of post-shutdown accident scenarios and fuel handling incidents in which the fuel at low decay power levels is postulated to heat up due to degraded cooling conditions. As a result, an appropriate test program funded by the Fuel and Physics Dept of the Technical and Training Services Division, was carried out.

The initial phase of the above test program involved the development of a test facility which is capable of oxidizing and simultaneously biaxially deforming Zr-4 fuel cladding samples. The test facility developed at OHRD consists of: a) in-house designed and built resistance tube furnaces

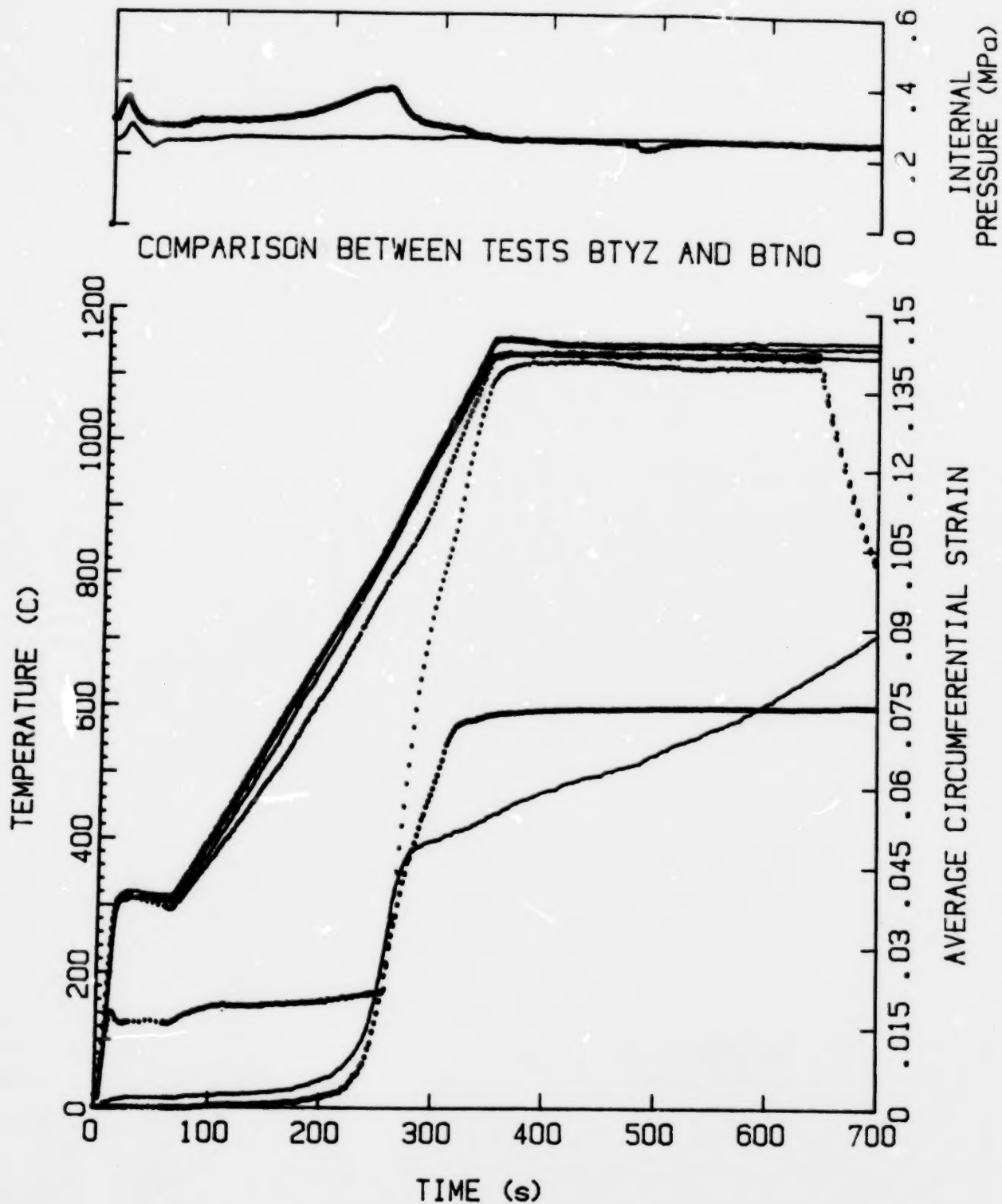


FIG. 1 PLOTS ILLUSTRATING THE DIFFERENCE IN DEFORMATION BEHAVIOUR BETWEEN TESTING IN ARGON/ARGON (SOLID CURVES) AND STEAM/ARGON ID/OD ENVIRONMENTS.

for specimen heating under closed-loop control; b) an in-house designed and built pressurizing circuit for biaxial deformation; c) a microprocessor-based data logger for data acquisition; d) an elapsed time indicator for accurate monitoring of time to failure in testing under biaxial conditions. Analysis of typical test facility component performance data obtainable indicates that a large variety of oxidation and elevated temperature biaxial tests may be performed using the newly developed facility.

The main part of the test program entails the generation of appropriate oxidation and biaxial deformation data. Results obtained indicate that braze heat-affected zone (HAZ) material oxidizes 1.2-2.0 times more rapidly than parent cladding material for temperatures in the range 600-850°C. Empirical correlations for the oxidation behaviour of both parent and braze HAZ materials were developed using the test data. For the test conditions examined, the oxidation of both parent and braze HAZ materials was consistent with post-transition or linear kinetic behaviour. The elevated temperature biaxial deformation behaviour was characterized by local ballooning in the weaker parent material at high pressures and by specimen failure in braze HAZ material at low pressures, due to the more rapid oxidation of the latter material. Tentative criteria for embrittlement and "handling" of partially oxidized CANDU Zr-4 fuel cladding were developed using the experimental oxidation data and the room temperature absorbed energy to fracture data. The tentative criteria for embrittlement and handling are presented in Figure 2, together with the current NSSD embrittlement criterion. Comparison between the newly developed tentative embrittlement criterion and the current NSSD embrittlement criterion indicates that the present criterion is conservative for temperatures above about 727°C when applied to parent fuel cladding material.

3.4.4 High Temperature Deformation Laboratory

In the past year, we increased the capability of the computer-controlled Materials Characterization Test Facility (MCTF) significantly by the acquisitions of a split, three-zone resistance furnace and a high resolution Zygo Laser Telemetric System for strain measurement. The acquisition of the resistance furnace will enable us to perform elevated temperature testing in situations where self-resistance of the test specimens is inappropriate (such as in the testing of ASME SA106B piping material at temperatures in the range 150-300°C). The acquisition of the high resolution laser-based strain measurement system will enable us to obtain more detailed stress-strain data in the critical micro-yielding regions of the flow curves.

Several new application programs for test control, data acquisition, data analysis and data plotting were developed for use with the computer-controlled MCTF.

3.4.5 Materials Properties for CANDU Reactor Ex-Core Component Alloys

Materials Properties for Darlington PHT Piping Material

An extensive tensile test program designed to generate mechanical properties data for ASME SA106 Grade B piping and SA105 forging materials has been carried out as part of the Darlington Leak-Before-Break Materials Test Program. Results obtained indicate that the parent piping material exhibits typical deformation behaviour for medium-carbon steels: that is, well-defined yield points, Luders or stretcher strains (lower yield plateaus) and some strain rate sensitivity. The results also showed that the tensile strength of the parent piping material exhibits a local minimum and maximum with temperature over the temperature range 20-350°C. In addition, it

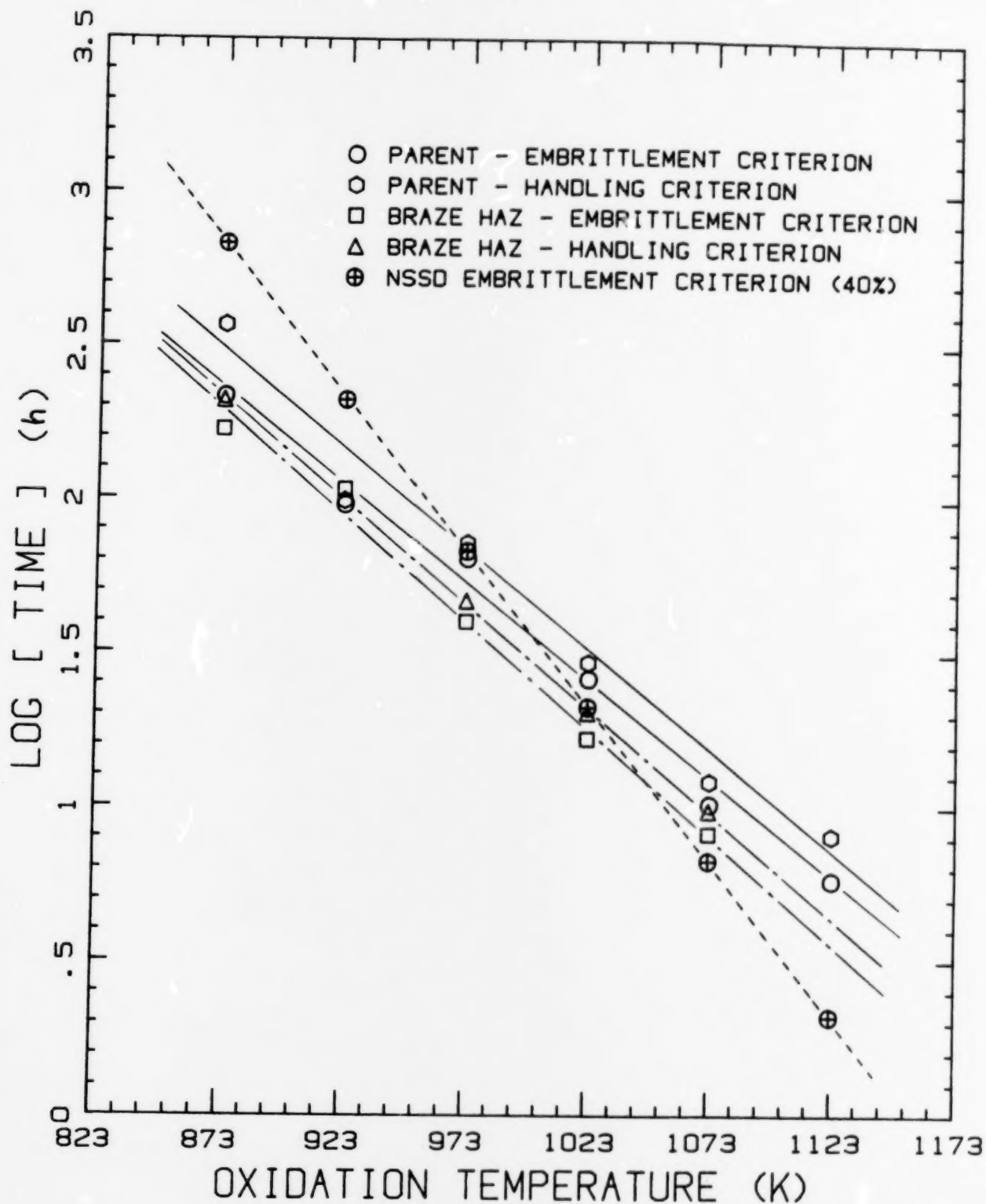


FIG. 2 PLOTS ILLUSTRATING THE VARIATION OF THE TENTATIVE CRITERIA FOR EMBRITTLEMENT AND "HANDLING", AND THE CURRENT NSSD EMBRITTLEMENT CRITERION FOR ZIRCALOY-4 FUEL CLADDING MATERIAL AS A FUNCTION OF TEMPERATURE.

was observed that test specimens containing weld and weld HAZ metal in their gauge lengths exhibited multiple yielding. This multiple yielding behaviour may be attributed to the presence of different microstructural zones, which yield at different stress levels, in the gauge length of the samples. Comparisons between longitudinal and circumferential results indicate that the piping material has approximately isotropic tensile strengths.

Close scrutiny of all test data indicates that, for purposes of analyses, the variety of SA106 piping and SA105 vessel forging materials studied may be grouped into the following categories: a) 22" and 24" NPS sch 100 pipes; b) vessel material; c) 12" NPS sch 100 pipes; d) weld and weld HAZ metal.

A review of the technical literature has been carried out (in collaboration with O.E. Lepik of the Nondestructive and Fracture Evaluation Section) to assess the effects of strain aging and related phenomena on the toughness of carbon and carbon-manganese weld metals. This review was motivated by some test results obtained from the Darlington Leak-Before-Break Materials Test Program which indicated that some non-stress-relieved welds in SA106 Grade B piping material exhibited lower than expected toughness at 250°C and cleavage fracture at room temperature. The factors reviewed are strain aging embrittlement, residual stresses, dilution of weld metal and quench aging. The effect of post-weld heat treatment was also reviewed since such action is known to improve weld toughness and reduce the risk of brittle failure. For completeness, the micromechanism of cleavage fracture in steels, and the effects of microstructure and the various factors reviewed on the material's resistance to cleavage fracture, were examined.

Information obtained from the literature review suggests that the reduction of fracture toughness observed in some Darlington piping weld material may be attributed largely to a strain aging effect. However, since PWHT of the PN-229 weld metal did not completely eliminate the initiation of localized cleavage in some small areas of the weld microstructure, it was concluded that other factors had contributed or interacted to cause the observed cleavage fracture.

3.5 Pressure Tube Development

The tubes manufactured by the specification recommended by Task Group 3 continue to be evaluated. A total of nine tubes (three from each production route) have been installed in Bruce 8. The in-service performance of these tubes will be monitored and evaluated to establish if such tubes are suitable for future LSFCR and if so which production route should be used.

There was some involvement with the evaluation of alternative pressure tubes manufacturers as well as with procedures used to melt Zr-2.5 wt% Nb ingots.

3.6 Corrosion-Fatigue

3.6.1 EPRI Project

This project was finally given the go-ahead just before year-end, roughly 1.5 years after the decision to award the contract to Ontario Hydro. Throughout the year, however, work continued on construction of the test facility so that now the contract is underway, actual testing should be possible almost immediately. Since development of the test facility and methodology was originally expected to take the first year of the contract period, the delay in formalizing the arrangements has clearly worked to EPRI's favour but, more importantly, should enhance the overall value of the project.

3.6.2 Corrosion Fatigue-Crack Propagation

This thermal plant research program was formulated for the generation of relevant crack growth data under simulated economizer operating conditions. Due to equipment problems, crack growth tests at 300°C and 16 MPa were not completed. The leakage at the sliding seal of the autoclave system has not been remedied, and the University of Toronto project team has been trying to improve the high pressure autoclave system using their own funding. A preliminary proof test at 5.2 MPa and 260°C was successful, however, whether the results could be related to the more severe service conditions present in the economizer header was not resolved. This delay in the program was reported to the Canadian Electrical Association and an extension was granted.

3.6.3 Corrosion-Fatigue of Carbon Steel Components in Generating Units

Corrosion-fatigue of boiler tubes is a major cause of forced outages in thermal units and similar damage in economizer headers has necessitated some costly replacements. In CANDU nuclear units, lack of relevant corrosion-fatigue crack growth rate data hinders the assessment of waterside flaws and, quite recently, a forced outage at Bruce NGS was attributed to corrosion fatigue of a carbon steel pipe. A major report was issued characterizing corrosion-fatigue damage in carbon steel, reviewing information on crack initiation and growth, identifying research needed to meet Ontario Hydro's present and anticipated needs and, finally, outlining some Research Division initiatives in this area. A recommendation was made to formulate a test program relevant to carbon steel components in the CANDU units.

3.6.4 Corrosion-Fatigue of Pressure Tube Material

A joint proposal was issued in collaboration with the Fracture Mechanics unit for an initial study of fatigue crack growth in CANDU pressure tube material, both in air and in primary heat transport water conditions. The tests in the water environment would use the autoclave - flow loop facility and would be carried out on both compact tension and four-point bend specimens. After some delay, a positive response to the proposal was received from CNS/NSD and work was in progress at year's end to adapt the facility for the work required.

3.7 Thermal Plant Life Assessment

As part of a general increasing effort related to remaining life assessment for thermal units, residual life estimations were carried out on large numbers of superheater and reheater tube sections from units at Lakeview, Lambton and Nanticoke. In some cases, predicted remaining lives were several hundred thousand hours so that the tubes would be expected to last the life of the unit. In other cases, however, lives of less than 20,000 hrs were predicted, indicating a need for short-term action. Following the work in 1985 to refine the life prediction model based on tube wall thinning, gradual development in the methodology took place during these investigations. As well as accounting for a creep contribution to wall thinning, where appropriate, use of steamside scale thickness measurement as a means of determining effective tube metal temperature became a standard procedure. Two models are available relating to scale thickness to metal temperature, one assuming parabolic growth kinetics, the other based on linear growth. At this point, the relative validity of these approaches remains to be determined, but some indications were gained from use of both methods during the various life assessments throughout the year.

3.8 Boiler Tube Failure Investigations

During the year, about a dozen investigations were conducted, some being reported by memo in the interests of expediency. Various types of damage and failure mechanisms were encountered eg, overheating, use of wrong material, cyclic system stressing, sootblower erosion, corrosion-fatigue and creep. In the investigation of a failed dissimilar metal weld in a Lambton superheater tube, an analytical approach involving the von Mises stationary state creep equivalent stress indicated that a modest bending stress had been effective in reducing the predicted life of the weldment by ~45%. It was recommended that other similar transition welds be inspected for cracking and, indeed, at year's end, damage was found in another Lambton unit. Incidentally, based on the estimated metal temperature, roughly 50% of the predicted creep life of the ferritic tube away from the weld had been expended at the time of the weld failure.

4.0 1987 PROGRAMS

4.1 In-Reactor Deformation

4.1.1 Analysis of Power Reactor Dimensional Changes

Plant measurements of pressure tube dimensional changes such as elongation, diametral expansion and sag are made periodically. Analysis of existing data will continue and analysis of new data will be on-going as data becomes available. It is anticipated that this study will identify trends in deformation behaviour which can then be rationalized in terms of reactor operating conditions and/or tube metallurgical parameters.

4.1.2 Irradiation Growth

Irradiation growth experiments on pressure tube material will continue at AERE Harwell. Samples irradiated at 280°C are now exceeding current Pickering NGS fluence. These samples will continue to be irradiated at 280°C until expected end-of-life fluences are attained. Experiments at an irradiation temperature of ~310°C will continue, but those at ~350°C will not. Those samples previously irradiated at 350°C will be irradiated at 280°C or 310°C. These tests have identified transient effects resulting from relatively small temperature changes which may have important implications to load maneuvering.

4.1.3 Irradiation Creep

It is planned that irradiation creep experiments on small tubes will continue at AERE Harwell. It is anticipated that the results of this study will allow more detailed experiments to be designed.

4.2 Out-Reactor Deformation

4.2.1 Study of Anisotropic Creep

The study to develop a procedure to predict anisotropic creep from compression creep tests will continue. The immediate task is to resolve the problems of load axiality, and sample buckling and barreling. Once these problems are overcome, testing of Zr-2.5 wt% Nb pressure tube material will commence.

4.3 Hydrides in Pressure Tubes

4.3.1 TRAC Program

This program is expected to continue throughout the year with the financial administration being completely performed by NSD. This will provide more time for more effective technical administration. The key areas of work involve getting more information on the nature of the thermal boundary conditions, determining the threshold deuterium levels for blister growth more precisely, showing that the modeling can be used as a predictive tool, and, finally, getting a better handle on the parameters which influence blister fracture.

The main direction of the technical effort will be towards the blister fracture area with a key experiment being the growth of blisters under stress to fracture. A description of the total blister fracture program for this year has already been outlined¹. Secondary experiments involve determining the effect of blister shape, hydride distribution, and thermal cycles on the stress to fracture blisters.

In the blister growth area, there are a number of issues to be resolved to determine the threshold hydrogen level required to form blisters in pressure tubes. In particular, the values for the heat of transport obtained in recent studies carried out under contract at the University of Toronto need to be reviewed and assessed for their implications. Effects of the hysteresis in the terminal solid solubility of hydrogen also need study.

4.3.2 Reactor Assessment Studies

Further development of the Monte Carlo model is required: NSD has already recognized this requirement. The timing of this development is currently in question since it does depend now, to a large degree, on the blister growth modeling task of the TRAC program. There have already been requests to apply the model already developed to assessments of a CANDU-600 unit and it is expected that there will be a demand for further assessments using the current model.

4.3.3 Crack Initiation in Pressure Tube Materials

The main objective is to determine the crack initiation criteria under increasing hydrogen and end-of-life service conditions. Further development of the exponential life description of the cantilever beam data will be carried out, in particular, the role of stress intensity factor and data scatter on life prediction will be studied in-depth. The in-house four point bend tests with increasing hydrogen condition and notched surfaces will continue. The new experimental results will help to clarify/confirm the observed relationship between the stress state and the time to crack nucleation.

4.3.4 Rolled Joint Integrity

The performance of low residual stress zero clearance rolled joint with high hydrogen content will be investigated. The main objective is to establish a relationship between the behaviour of

1. Memorandum: M. Leger and A.C. Wallace to D.W. Souther, TRAC Task 415.1-Crack Initiation at Blisters, Proposed 1986 Program at Ontario Hydro Research Division, File: 823.21 (TRAC), September 12, 1986.

small specimens and the performance of a full sized components. Small defects will be machined into the ID surface at locations of high residual stress. The assembly will then be subjected to both thermal and pressure cycling. Both the microstructure and hydride distribution after testing will be compared to that observed in small specimens exposed to similar test conditions.

4.3.5 Load Following

Most of the effort will be devoted to the completion of a scoping test matrix to determine the worst possible damage with combined thermal and mechanical cycling to the test specimens. The role of residual stress after straightening of as received pressure tube will also be examined, the tensile test results will be compared to that obtained from curved tensile specimens which did not see any mechanical straightening.

4.3.6 P8 Fretted Tube Hydrogen Diffusion

The program is expected to be completed by the end of the first quarter with a reasonable understanding of the hydrogen distribution in fretted tubes after long term exposure to simulated reactor temperature and pressure. Prediction of the hydrogen profile from theoretical modelling will be carried out. The second fretted tube will be removed from the furnace and the hydrogen redistribution measured. The results will be compared to the receding interface model found applicable to the first fretted tube.

4.3.7 DHC Growth Studies

Work proposed in 1987 will address the following:

1. Determine how crack velocity is affected by strength of pressure tubes.
2. Determine the effects of depressurization and load cycling on DHC behaviour.
3. Evaluate the effectiveness of temperature maneuver in slowing down crack growth.
4. Obtain crack growth data at outlet temperature under operating thermal cycles.
5. Determine the maximum cracking temperature as a function of hydrogen concentration and strength of pressure tubes.

4.3.8 Fracture Toughness Evaluation

Another Zr-2.5 Nb pressure tube section will be hydrided under internal pressure to obtain fully radial hydrides. This along with the tube containing mixed hydrides will be tested by the Fracture Mechanics Unit to determine the ductile to brittle transition temperature. Post fracture examination of the specimens and determination of the fracture mechanisms will be performed by the Materials Integrity Section.

4.3.9 Crack Initiation from Pickering 8 Tubes Containing Fretted Marks

Tests performed in 1986 showed that delayed hydride cracking can be initiated from deep simulated secondary fretted marks under twice the operating hoop stress. This program will be

continued to determine whether cracks can be initiated under reactor operating stress. Further tests will also be performed to ensure that the value of the applied stress is correct.

4.3.10 Detection of Hydrides in Pressure Tubes

The programs proposed for 1987 are a continuation of 1986 programs. A rigorous investigation of resistivity and thermoelectric effects of hydrogen on small samples will be performed. If these results are favourable, then large scale tests upon actual pressure tube sections will be initiated. This will entail the construction of a test head which can travel down the pressure tube interior. Calorimetric investigation of hydrogen in Zr - 2.5 wt% Nb material will be performed in greater detail. A continuing search for techniques for hydrogen determination in pressure tube material is also ongoing.

4.3.11 Committee Participation

The level of participation can probably be reduced in the coming year as soon as the REFAB Assessment report is completed (expected at the end of February). There will be some involvement in consulting on the Hydrogen Sinks for Darlington Project which will also be heaviest at the beginning of the year.

4.4 Transmission Electron Microscopy

4.4.1 Dislocation Substructures in Zr-2.5 Wt% Nb Pressure Tube Alloys

T.E.M. studies of dislocation substructure in Zr-2.5 Nb pressure tube alloys will be continued with a view to understanding their role on the mechanical properties and irradiation growth response of the alloy. The main emphasis will be concentrated on the more quantitative analysis of the α/β interface structure as a result of the β_{Zr} (b.c.c.) $\rightarrow$ α_{Zr} (h.c.p.) phase transformation. It is expected that these results as well as the study on the α/α -low angle boundaries will soon be completed to the point for an external publication.

The first T.E.M. characterization of the Zr-2.5 wt% Nb pressure tubes extruded at much lower extrusion ratio (4:1 as opposed to 7.8:1 for the TG3 pressure tubes and 11:1 for the current pressure tubes) will be performed. The slices of the axial, transverse and radial sections after different hold points are already prepared and the first T.E.M. examinations are expected to start very soon.

4.4.2 Hydride Precipitation

Further T.E.M. studies on the role of the residual internal stresses on hydride precipitation in Zircaloy-2 swaged rod will be continued. The research is intended to have three objectives: (i) to characterize in more detail transgranular hydrides in the 4% tensile specimens, (ii) to study hydride distribution in both tensile and compressive specimens with higher hydrogen levels, (iii) to attempt to assess the role of twinning on hydride precipitation. The stress state in the compressive specimen when the major mode of deformation is twinning is not known at present. The hydride distribution will be studied using optical and electron metallographic techniques and the residual stresses will be measured by the neutron diffraction technique. Development of the

dislocation substructure in the same material subjected to various mechanical treatments (tension, compression and cycling) will also be studied. It is believed that the development of the characteristic dislocation substructure after various mechanical treatments, is closely related to the residual internal stresses.

Studies on stability of the microstructure in the Zr-2.5 wt% Nb pressure tubes and its role on hydride distribution will be continued as input to the program on delayed hydride cracking.

4.4.3 T.E.M. Studies of Irradiated Zr-2.5 Wt% Nb Pressure Tubes

As reported, the dislocation substructure in the unirradiated Zr-2.5 wt% Nb pressure tubes is composed of dislocations with different Burgers vectors. The presence of different types of dislocations and their spatial distribution should influence irradiation growth by altering the manner in which irradiation produced point defects are partitioned among various sinks. A number of samples of various Zr-2.5 wt% Nb pressure tube materials have been irradiated in the reactor at Harwell. T.E.M. examination of these samples should provide us with valuable information on the effect of neutron irradiation on dislocation substructure, phase stability and the nature of neutron damage in the pressure tube materials.

4.4.4 Dislocation Substructure in Zircaloy Pressure Tube Alloy

T.E.M. characterization of dislocation substructure in the Zircaloy pressure tube alloy will be continued. This study is associated with a program on the mechanical properties of the Zircaloy pressure tube material which is in progress to develop a better understanding of deformation mechanism in zirconium.

4.4.5 T.E.M. Investigation of the Bainite Transformations IN Fe-Ni-C, Fe-Si-C and Fe-Si-Mo-C Alloys

It is proposed to study the morphologies and substructures of bainitic ferrite formed isothermally in Fe-Ni-C, Fe-Si-C and Fe-Si-Mo-C alloys by means of electron microscopy. These observations together with kinetic data for bainitic ferrite growth will be interpreted in terms of ideas concerning the structure and migration characteristics of f.c.c./b.c.c interfaces. These studies will be done in collaboration with Prof. G.R. Purdy, McMaster University. These studies are considered a good educational opportunity which will expand knowledge in phase transformations as well as provide an opportunity to perform research with advanced analytical electron microscopes at McMaster and Toronto Universities.

4.5 Materials Properties for CANDU Reactor Core-Component Alloys

4.5.1 Fundamental Studies of Elevated Temperature Deformation Behaviour

The advent of powerful computer codes for use in the flaw and materials integrity evaluation of critical engineering components, such as the pressure tubes in CANDU reactors, requires detailed information on the flow behaviour of the component material. To augment the stress-strain data base available for Zr-2.5 Nb PT material, a series of tensile tests over the temperature range 100-500°C using a newly acquired high resolution Zygo Laser Telemetric System for strain measurement will be performed. The use of this laser-based strain measurement system will enable us to characterize more fully the initial regions of the stress-strain curves.

To fully assess the effects of proposed load-manoeuvring operations on the performance of CANDU reactors, new information is required. Such new information includes the thermal creep behaviour of Zr-2.5 Nb pressure tubes under cyclic temperature conditions. Therefore, it is proposed that uniaxial and biaxial creep tests will be carried out under cyclic temperature conditions. The precise test conditions will be determined after consultation with the appropriate Design and Development - Generation Division personnel.

4.5.2 Materials Properties for Zircaloy-4 Fuel Cladding

In 1987, it is planned to carry out additional oxidation and mechanical tests to increase the data base for CANDU reactor Zircaloy-4 fuel cladding material. Such data are needed to develop more quantitative criteria for embrittlement and handling of partially oxidized fuel cladding. The above criteria are needed in the analysis of a number of post-shutdown accident scenarios and fuel handling incidents in which the fuel at low decay power levels is postulated to heat-up due to degraded cooling conditions.

Consultation will be provided to other programs which require expertise on the materials properties of Zircaloy-4 fuel cladding material. The appropriate projects will be undertaken when the need arises.

4.6 Materials Properties for CANDU Reactor Ex-Core Component Alloys

4.6.1 Darlington PHT Piping Material

A test program (described in OHRD Report M86-158-K) has been proposed to study the effect of post-weld heat treatment on the toughness of weldments in SA106B piping material. This program will be carried out in collaboration with O.E. Lepik of the Nondestructive and Fracture Evaluation Section and M.A. Clark of the Metallurgy Section. The Materials Integrity Section will perform tensile testing to characterize the serrated flow (dynamic strain aging) behaviour of the weld material.

4.6.2 General Materials Testing

Over past few years a very powerful computer-controlled MCTF and several experimental procedures for use in the High Temperature Deformation Laboratory have been developed. Although the above-mentioned MCTF and test techniques were developed for use in the elevated temperature testing of CANDU core-component materials, the developed hardware and software can easily be modified for testing of other materials, at both ambient and elevated temperatures. Therefore, in 1987 it is proposed to explore new applications for the MCTF.

4.7 Pressure Tube Development

4.7.1 Pilgered Pressure Tubes

Texture, grain size, microstructure and hydride reorientation stress of the pilgered tubes will be evaluated and compared to the current cold drawn tubes.

4.8 Corrosion-Fatigue

4.8.1 Corrosion Fatigue-Crack Propagation

The University of Toronto project team will continue with their own funding to resolve the equipment problems. Limited crack growth data will be generated, and a meeting with the Canadian Electrical Association project manager to discuss the lack of progress and future direction of the study will be arranged.

4.8.2 Corrosion Fatigue-Crack Initiation

The objectives of this EPRI funded project are to determine the root cause of corrosion fatigue in thermal plant boiler tubes and to find solutions to the problem. The role of OHRD is to simulate the environmental cracking found in waterwall tubes and to determine the variables that affect crack nucleation. Using this developed laboratory technique, OHRD will complete a test matrix based on input from other workers in the project (field monitoring, stress analysis).

The major task will be the design and validation of a tubular specimen to be tested in a simulated boiler environment. Most of the components of a flowing high pressure high temperature loop have been ordered and assembled. A commissioning test will be performed to evaluate each component in the loop. The capability to control the flow and the water chemistry will be the critical area for assessment. The tubular specimen will be fatigue loaded to determine if realistic corrosion fatigue cracking can be produced in such a specimen geometry when exposed to the simulated boiler water.

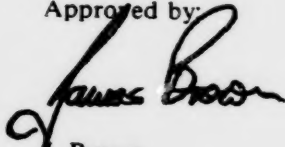
To determine the onset of crack initiation, a nondestructive method to monitor the specimen continuously will be required. The specimen's compliance may provide an early indication of change in the specimen's deformation and hence the onset of crack nucleation and growth. The potential drop method is also a well known method for the accurate detection of cracking in standard Fracture Mechanics specimens. Several trial runs and an iterative process will be required to lead to the final selection of a monitoring procedure.

5.0 SUMMARY

The work load on the Section Staff will remain high throughout 1987. Studies on pressure tubes are largely co-ordinated through the joint OH-AECL Fuel Channel Research Program. Several of the studies will be funded under the CANDU Owners Group (COG) agreement and this will result in an increased involvement with AECL-RC programs along with continued up-date of the schedules etc, of the current studies. The section also provides a general consultation service to various Ontario Hydro Departments and these relatively short-term commitments can result in significant manpower requirements when considered over the complete year. The Section staff involvement with pressure tubes covers the alloy production, tube manufacture, properties of

finished product, in-service performance and predictions of end-of-life properties. The staff resources available to the Materials Integrity Section are still considered to be below minimum level to provide the short-term requests and still continue the longer-term programs within agreed to schedules.

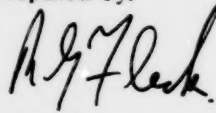
Approved by:



J. Brown
Manager
Metallurgical Research Department

RGF:ljc

Prepared by:



R.G. Fleck
Supervising Engineer
Materials Integrity Section

REPORTS

- 86-11-K Tensile Properties of ASTM SA106 GR B & SA105 Darlington Primary Heat Transport Piping Materials at 293 & 523 K, January 14, 1986
E.T.C. Ho and M.J. McGraw
- 86-39-K Materials Integrity Section - A Review of 1985 and Proposals for 1986, February 10, 1986
R.G. Fleck
- 86-52-K Residual Life of Superheater and Reheater Tubes at Nanticoke Generating Station, Unit 5, April 4, 1986
J.D. Parker
- 86-53-K Effect of Straightening Operation on the Transverse Tensile Properties of Zircaloy-2 and Zr-2.5 Wt% Nb Pressure Tube Materials, March 18, 1986
E.T.C. Ho and M.J. McGraw
- 86-60-K Correlation of Pressure Tube Elongation With Materials Microstructure, April 16, 1986
J.M. Tadros and J.D. Parker
- 86-71-H Corrosion-Fatigue Steel Components in Generating Units, May 2, 1986
H.J. Westwood
- 86-93-K The Effect of Hydride Morphology on the Axial Fracture Toughness of Zr-2.5 Nb Pressure Tube Material, July 15, 1986, O.E. Lepik, G.K. Shek, A.C. Wallace, C.P. Stearns, H. Seahra.
- 86-135-H Residual Life Estimation for Lakeview Unit 4 Reheater Tubes, May 29, 1986
H.J. Westwood
- 86-153-K A Study on Hydride Blister Formation in Bruce 3 and 4 and Pickering 5 and 6, November 11, 1986
M. Leger and D. Leemans
- 86-173-K Biaxial Deformation Behaviour of Zr-2.5 Wt% Nb Pressure Tubes in Various ID/OD Environments at Low Internal Pressures and Elevated Temperatures, October 2, 1986
E.T.C. Ho
- 86-198-K Deformation Behaviour of Zr-2.5 Wt% Nb Pressure Tube Material in an Air Environment in the Temperature Range 1473-1773 K, December 2, 1986
E.T.C. Ho
- 86-211-H Damage Assessment and Residual Life Estimation of Reheater Tubing - Nanticoke GS Unit 5, September 4, 1986
H.J. Westwood

- 86-323-K "Effects of Strain Aging and Related Phenomena Upon the Toughness of Carbon and Carbon-Manganese Weld Metals - A Literature Review", E.T.C. Ho and O.E. Lepik.
- 86-342-K "An Experimental Facility for Oxidation and Biaxial Deformation Studies of Zircaloy-4 Fuel Cladding Samples", E.T.C. Ho and A. Donner.
- M85-139-K Detailed Flow Curves for Darlington Primary Heat Transport Piping Material at 293 and 523 K, June 13, 1986
E.T.C. Ho and M.J. McGraw
- M86-28-K Metallurgical Examination of Rear Waterwall Tube - Nanticoke GS Unit 7, March 19, 1986
H.J. Westwood
- M86-54-K Metallurgical Examination of failed Reheater Hairpin Bend - Lakeview GS #2, June 19, 1986
H.J. Westwood
- M86-57-K Visit Report: International Conference on Creep, Nihon Toshi, Tokyo from April 13-18, 1986, and Visits to Japanese Industry, June 5, 1986
J.D. Parker
- M86-59-K Downcomer Equalizing Line Failure - Lakeview TGS Unit 2, May 28, 1986
H.J. Westwood
- M86-60-K Examination of Thunder Bay Unit 2 Reheater Tubing, June 4, 1986
H.J. Westwood
- M86-63-K Residual Life Estimation - Lakeview Unit 3 Reheater Tube Samples, June 5, 1986
H.J. Westwood
- M86-64-K Failure Analysis of Waterwall Tube - Lakeview GS Unit 1, June 19, 1986
H.J. Westwood
- M86-76-K Examination of Waterwall Tube Samples, Nanticoke TGS, Unit 3, July 25, 1986
H.J. Westwood
- M86-82-K Superheater Header Drain Line Failure - Lakeview GS Unit 2, July 25, 1986
H.J. Westwood
- M86-89-K Lakeview Unit 6 Waterwall Tube Failures - Metallurgical Investigation, September 5, 1986
H.J. Westwood
- M86-96-K Data Logging Using a Hewlett Packard 3497 A Data Acquisition/Control Unit, September 8, 1986
D.A. Oad

- M86-99-K Reheater Tube Residual Life Assessment - Lambton TGS Unit 1.
September 29, 1986
H.J. Westwood
- M86-103-H Analysis of Dissimilar Metal Pressure Weld Failure - Lambton TGS - Unit 3.
October 16, 1986
H.J. Westwood
- M86-108-K Interim Report on Hydrogen Diffusion Tests on P-8 Fretted Tubes.
November 6, 1986
W.K. Lee
- M86-131-K Delayed Hydride Cracking Behaviour in Zr-2.5 Nb Under Simulated
Depressurized Condition, December 19, 1986
G.K. Shek and H. Seahra
- M86-158-K "Proposal to Evaluate the Effect of Post-Weld Heat Treatment on Toughness of
Weldments in SA106B Piping Material", E.T.C. Ho, O.E. Lepik and M.A. Clark.

PUBLICATIONS

Life Prediction of Cylindrical Components Operating Under High Temperature Creep Conditions. EPRI Conference on Life Extension and Assessment of Fossil Plants. June 3, 1986. J.D. Parker

Corrosion-Fatigue Cracking in Fossil-Fueled Boilers by H.J. Westwood and W.K. Lee.

Proceedings of International Conference on Fatigue, Corrosion Cracking, Fracture Mechanics and Failure Analysis, Salt Lake City, Dec 1985, published in 1986.

Hydride Blister Growth in Zr-2.5 wt% Nb Pressure Tube Material. Presented to 12th Annual Symposium on Simulation of Reactor Dynamics and Plant Control, Canadian Nuclear Society, Hamilton. April 21, 1986. M. Leger and T.P. Byrne.

PRESENTATIONS

Hydride Blister Growth in Zr-2.5 wt% Nb Pressure Tube Material. Presented to 12th Annual Symposium on Simulation of Reactor Dynamics and Plant Control, Canadian Nuclear Society, Hamilton. April 21, 1986. M. Leger and T.P. Byrne.

J.D. MacArthur, T. Laursen, Xin-Pei Ma, G.R. Palmer, J.L. Whitton and M. Leger, Deuterium Concentration Distribution in Blisters at Cooled Regions of CANDU Reactor Type Pressure Tubes, Presentation by J.D. MacArthur to International Meeting on Nuclear Analytic Methods, Orleans, France, June 1986.

Ontario Hydro Experience with Dissimilar Metal Welds and Residual Life of Headers and Steam Pipes. Seminar with Japanese Study Mission on Residual Life Assessment. June 9, 1986. J.D. Parker

Residual Life Assessment of Steam Tubing and Corrosion-Fatigue of Boiler Tubing. Seminar with Japanese Study Mission on Residual Life Assessment. June 9, 1986. H.J. Westwood
Hydride Blister Growth in Zr-2.5 Wt% Nb, Presentation to Queen's University, Department of Physics, October 1986, M. Leger.

CONFERENCES AND TECHNICAL MEETINGS

Condition Monitoring of Steam Lines Using Strain Measurement - EPRI Meeting, March 17, 1986, J.D. Parker.

EPRI Meeting on Steam Line Inspection and Condition Monitoring, March 17, 1986, Palo Alto, California, J.D. Parker.

CRNL: Progress on Understanding Creep and Growth of Calandria Tubes, March 19, 1986, Chalk River, Ontario, J.D. Parker.

SLAR Assessment Committee, Technical Assessment of Spacer Location and Repositioning, Nuclear Systems Department, Design and Development Div Report No. 86-066, April 1986.

International Conference on Creep and Visits to National Research Institute for Metals, April 13-22, 1986, Nippon Kokan, Tokyo, J.D. Parker

EPRI Conference on Life Extension and Assessment of Fossil Plant, June 2-4, 1986, Washington, USA, J.D. Parker.

EPRI "Industry Advisory Group" Meeting on Remaining Life Assessment, June 5, 1986, J.D. Parker.

Canadian Metal Physics Conference, June 4-6, 1986, Kingston, Ontario, J.M. Tadros.

11th International Congress on Electron Microscopy, August 31-September 7, 1986, KYOTO, Japan, V. Perovic.

The Metallurgical Society Fall Meeting, October 5-9, 1986, Orlando, Florida, G.K. Shek.

Meeting on Fracture Program for CANDU Fuel Channel - CRNL, November 19-21, 1986, Chalk River, Ontario, G.K. Shek, W.K. Lee, V. Perovic, M. Leger, K. Tashiro.

DISTRIBUTION

Dr. D. Mills	Director of Research
Mr. M.J. Nichols	Tech & Train Serv, A7H1
Mr. B. Montford	Tech & Train Serv, A7H8
Dr. M.W. Shanahan	Tech & Train Serv, A7H3
Dr. P.T. Truant	Tech & Train Serv, A7A10
Mr. R.W. Patterson	Tech & Train Serv, A8H3
Dr. D. Sidey	Tech & Train Serv, A8H2
Mr. P.R. Burroughs	Design & Development - Gen, H15B1
Mr. D. Souther	Design & Development - Gen, H10A1
Dr. G.J. Field	Design & Development - Gen, H10A16
Mr. C. Poidevin	Design & Development - Gen, H10A8
Mr. R. Lior	Design & Development - Gen, H10D16
Mr. M. Hardie	Design & Development - Gen, H10C16
Dr. R.A. Brown	Design & Development - Gen, H9A27
Mr. A. Muzumdar	Design & Development - Gen, H9B14
Dr. G. Blahnik	Design & Development - Gen, H9A13
Mr. E.G. Price	AECL-CO
Dr. J.R. MacEwan	AECL-RC (CRNL)
Dr. B.A. Cheadle	AECL-RC (CRNL)
Dr. S.R. MacEwen	AECL-RC (CRNL)
Mr. R. Dutton	AECL-RC (WNRE)
Dr. L.A. Simpson	AECL-RC (WNRE)
Dr. J. Brown	Metallurgical Research
Dr. M. Leger	Metallurgical Research
Dr. E.T.C. Ho	Metallurgical Research
Dr. V. Perovic	Metallurgical Research
Mr. W.K. Lee	Metallurgical Research
Mr. G.K. Shek	Metallurgical Research
Dr. H.J. Westwood	Metallurgical Research
Miss J. Tadros	Metallurgical Research
Dr. K.M. Tashiro	Metallurgical Research
Dr. R.G. Fleck (2)	Metallurgical Research

Editorial Section (2)
Public Reference Centre
Office Copies (3)
Extra Copies (3)